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INSIGHTS FROM INVESTIGATIONS OF IN-VESSEL RETENTION FOR HIGH POWERED REACTORS

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ABSTRACT

In a three-year U.S. - Korean International Nuclear Energy Research Initiative (INERI), state-of-the-art analytical tools and key U.S. and Korean experimental facilities were used to explore two options, enhanced ERVC performance and the use of internal core catchers, that have the potential to increase the margin for in-vessel retention (IVR) in high power reactors (up to 1500 MWe). This increased margin has the potential to improve plant economics (owing to reduced regulatory requirements) and increase public acceptance (owing to reduced plant risk). Although this program focused upon the Korean Advanced Power Reactor - 1400 MWe (APR1400) design, recommendations were developed so that they can easily be applied to a wide range of existing and advanced reactor designs. This paper summarizes new data gained for evaluating the margin associated with various options investigated in this program. Insights from analyses completed with this data are also highlighted.

KEYWORDS

In-vessel retention, in-vessel core catcher, external reactor vessel cooling.

1. INTRODUCTION

In-vessel retention (IVR) of core melt is a key severe accident management strategy adopted by some operating nuclear power plants and proposed for some advanced light water reactors (ALWRs). For certain low probabability accident sequences where there is inadequate cooling during a reactor accident, a significant amount of core material could become molten and relocate to the lower head of

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the reactor vessel [as happened in the Three Mile Island Unit 2 (TMI-2) accident]. If it is possible to ensure that the lower head remains intact so that relocated core materials are retained within the vessel, the enhanced safety associated with these plants can reduce concerns about containment failure and associated risk. For example, the enhanced safety of the Westinghouse Advanced 600 MWe PWR (AP600), which relied upon External Reactor Vessel Cooling (ERVC) of the reactor vessel submerged in a containment cavity filled with water, resulted in the U.S. Nuclear Regulatory Commission (US NRC) approving the design without requiring certain conventional features common to existing LWRs. However, it is not clear that currently proposed ERVC without additional enhancements could provide sufficient heat removal for higher-power reactors (up to 1500 MWe). Hence, a three-year, U.S. - Korean INERI project was conducted in which the Idaho National Laboratory (INL), Seoul National University (SNU), Pennsylvania State University (PSU), and Korea Atomic Energy Research Institute (KAERI) explored options, such as enhanced ERVC performance and the use of in-vessel core catchers (IVCC) that have the potential to ensure that IVR is feasible for high power reactors (Rempe *et al.,* 2005).

The systematic approach applied in this project combined state-of-the-art analytical tools and key U.S. and Korean experimental facilities. Evaluations focussed on modifications to enhance ERVC (improved data, vessel coatings to enhance heat removal, and an improved vessel/insulation configuration to facilitate steam venting) and modifications to enhance in-vessel debris coolability (enhanced IVCC configuration, thickness, and material). Figure 1 identifies some of the key U.S. and Korean experimental facilities and state-of-the-art analytical tools that were applied to investigate options that could enhance ERVC and IVCC performance. Table 1 summarizes pheomena investigated by experimental facilities and analytical tools applied in this program.

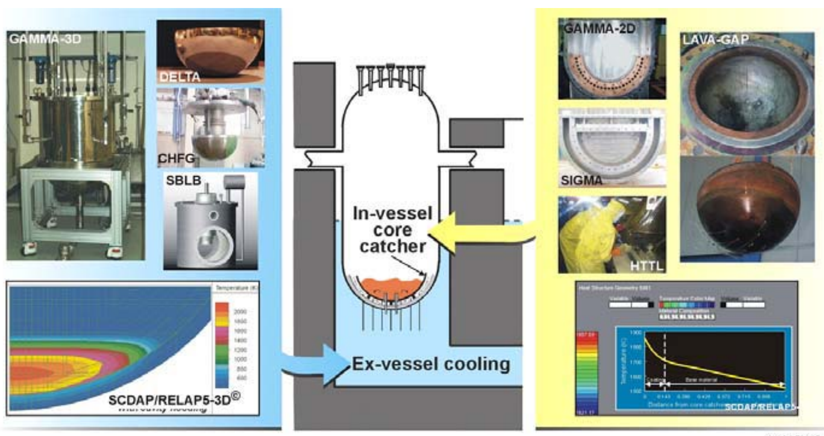


Figure 1: Key U.S. and Korean experimental facilities and state-of-the-art analytical tools applied to investigate options that could enhance ERVC and IVCC performance.

Phenomena	Name (Organization)
Experimental Facilities	
Narrow Gap Cooling	GAMMA 1D and 2D - Gap Apparatus Mitigating Melt Attack (SNU)
	DELTA 1D and 3D - Downward Ebullient Laminar Transition Apparatus (SNU)
	CHFG – Critical Heat Flux Gap (KAERI)
External Reactor Vessel Cooling	GAMMA 3D - Gap Apparatus Mitigating Melt Attack (SNU)
	HERMES-HALF (KAERI)
	SBLB – Subscale Boundary Layer Boiling (PSU)
In-Vessel Core Catcher Performance	LAVA-GAP (KAERI)
	HTTL – High Temperature Test Facility (INL)
Analytical Tools	
Plant Transient Evaluations	SCDAP/RELAP5-3D© (INL)
	SCDAP/RELAP5 MOD3.3 (KAERI)
Vessel Thermal Response	LILAC (KAERI)
	VESTA (INL)

Table 1: Experimental facilities and analytical tools applied to investigate IVR

As indicated in Figure 2, this three-year project included four tasks. In Task 1, calculations were conducted to define representative bounding late phase melt conditions. Characteristic parameters from those bounding conditions (thermal loads, pressure, relocated mass, etc.) were used to design an optimized in-vessel core catcher (IVCC) in Task 2 and ERVC enhancements in Task 3. In Task 4, collaborators assessed the improved margin obtained with Task 2 and 3 design modifications. As noted in Figure 2, this program focussed on the Korean Advanced Power Reactor -1400 MWe (APR1400) design (Oh, 2002). However, margins offered by each modification were evaluated such that results can easily be applied to a wide range of existing and advanced reactor designs.

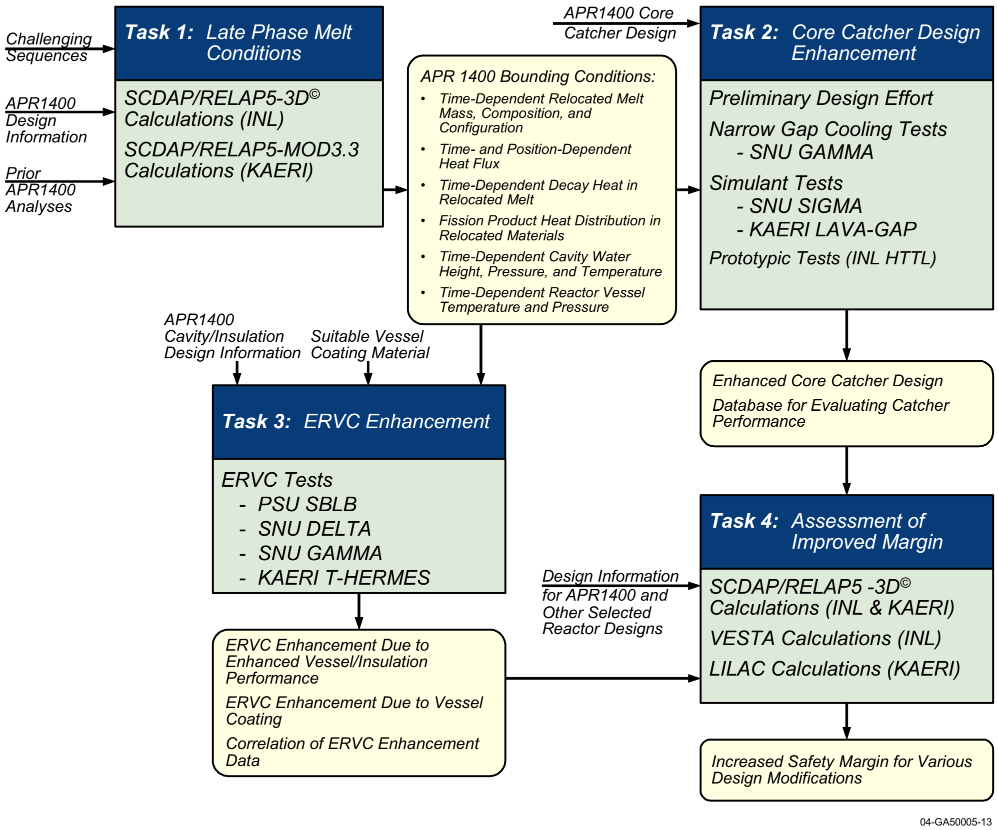


Figure 2: Project approach.

2. KEY PROGRAM ACCOMPLISHMENTS AND INSIGHTS

This paper summarizes key facilities applied and results obtained from this program. Insights gained from evaluating the margin associated with various options investigated in this program are also discussed. More detailed information about results obtained from each task may be found in (Rempe *et al.*, 2005).

2.1 Task 1 SCDAP/RELAP5-3D© and SCDAP/RELAP5MOD3.3 calculations

To obtain quantify representative late-phase melt conditions that could affect the potential for IVR of core melt following a severe accident in the APR1400, INEEL applied the SCDAP/RELAP5-3D© code (INEEL, 2003) and KAERI applied the SCDAP/RELAP5/MOD3.3 (Siefkin *et al.*, 2001) and SCDAP/RELAP5-3D© codes to the APR1400 plant. As indicated in Figure 3, fairly detailed plant models were developed using information from Korea Hydro and Nuclear Power Company, Ltd. (Oh, 2002) KAERI and INEEL models and input were similar with 250 RELAP volumes, 284 heat structures, and 316 junctions used to represent various components, structures, and piping in the APR1400 RCS.

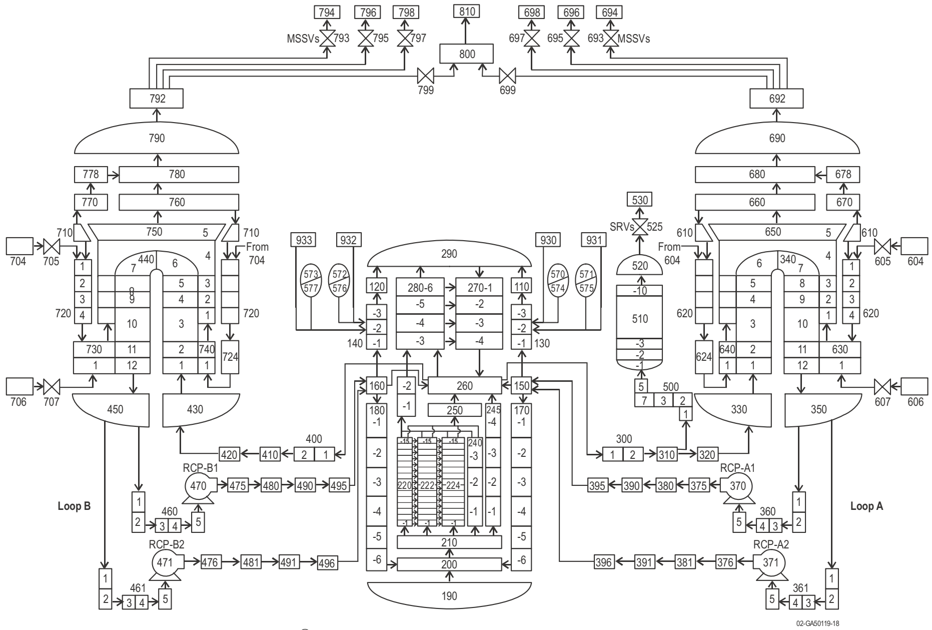


Figure 3: INEEL SCDAP/RELAP5-3D© model nodalization of APR1400.

Although an extensive series of severe accident calculations is required to identify bounding transients, Loss of Coolant Accidents (LOCAs), Station BlackOuts (SBOs), and Loss of FeedWater (LOFW) scenarios were assumed to be major IVR scenarios. Accordingly, a cold leg break (representing the LOCA response) and an SBO with LOFW (to combine remaining dominant IVR scenarios) were selected for analysis. Predicted values for vessel failure time, hydrogen generation, melt relocation masses, melt relocation volumes, decay heat in the relocated corium, and power densities in the relocated corium were compared. For the cases of interest, values predicted by SCDAP/RELAP5/MOD3.3 and SCDAP/RELAP5-3D© were similar.

Regardless of the transient considered, results for all calculations led to predictions of large melt masses (~100,000 kg total, or more) relocating at high temperatures (~3,000 K, or higher). These results appear to be consistent with the nature of the transients considered. Specifically, all cases involved complete core dryout and subsequent core heatup in a steam environment. Protracted periods (~1 h, or more) of complete core uncovery were sustained in each calculation, leading to development of large core melt masses at temperatures well above the fuel liquidus.

As an example, results for the LOCA analysis, which assumed a 0.0465 m^2 break in one of the cold legs in the primary coolant loop containing the pressurizer, predicted core melt and relocation into the lower head by 4,990 s (see Table 2). The timing of this relocation was earlier than all other transients analyzed (by as much as 6,110 s). Consequently, power retained in the melt was relatively high (because the decay period after reactor trip was relatively short). Hence, the decay power associated with this relocation yielded the highest thermal load for the APR1400 lower head.

Condition	Value
Relocation time, s	4990
Relocation Mass, kg	
UO_2	108,000
ZrO_2	5,180
Stainless Steel*	3,520
Total**	119,000
Decay Power, MW	51.2
Power Density, MW/m^3	3.48

*Task 4 calculations considered additional melting of structural material, so that the mass of stainless steel was 100,000 kg.
**Includes 276 kg Zr, 2,350 kg of control rod absorber material, and 95 kg of stainless steel from earlier relocations.

Table 2: Debris conditions for LOCA-1 at vessel failure

Estimated lower head average heat fluxes for the APR1400 transients considered ranged from 0.147 to 1.64 MW/m^2, which exceed peak lower head heat fluxes predicted for the Westinghouse AP600 reactor by factors as high as ~2.3 (Rempe *et al.,* 1997). Furthermore, the estimated APR1400 average heat fluxes exceed current estimates of the maximum CHF (for an uncoated vessel without an enhanced insulation design) by as much as a factor of ~1.2. These results indicate IVR may not be feasible without additional measures such as a core catcher and/or modifications to enhance ERVC.

2.2 Task 2 IVCC design development and evaluation

The approach adopted for developing an APR1400 IVCC design is illustrated in Figure 4. As shown in this figure, initial efforts focused on developing a preliminary in-vessel design. This was done at INEEL using a combination of scoping materials, flow, thermal, and structural analyses and scoping materials interaction tests. In addition, more detailed experimental data were obtained in two areas to support analysis of this IVCC. First, data were needed to estimate the heat that can be removed from the narrow "engineered" gap between the IVCC and the inner surface of the reactor vessel. As indicated in Figure 4, data were obtained from the GAMMA (Gap Apparatus Mitigating Melt Attack) facilities at SNU and the CHFG (Critical Heat Flux Gap) facility at KAERI to formulate a complete "narrow gap" boiling curve. Second, data were needed to understand the heat loads to the core catcher and demonstrate the viability of materials proposed for the IVCC. As illustrated in Figure 4, these needs were addressed by conducting tests in several facilities: the SIGMA (Simulant Internal Gravitated Material Apparatus) facilities at SNU were used to develop natural convection heat transfer correlations, the LAVA-GAP facility at KAERI was used to assess the impact of the IVCC on thermal heat loads to the vessel, and INEEL's HTTL (High Temperature Test Laboratory) was used to assess the potential for materials interactions.

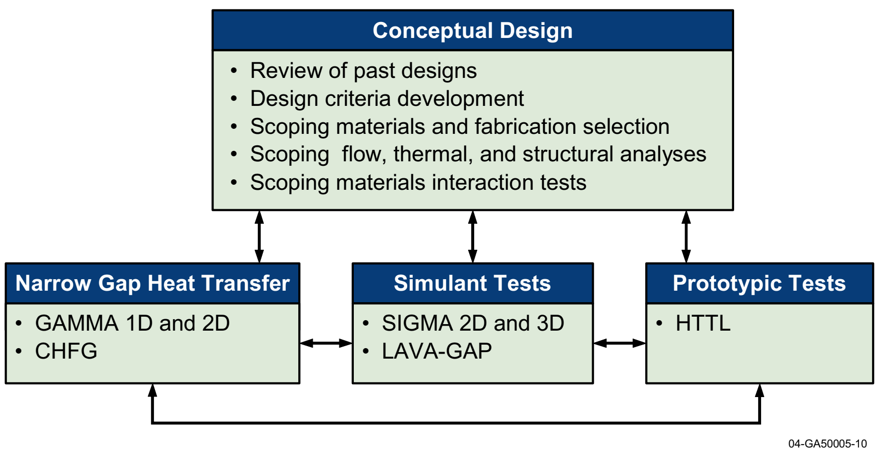


Figure 4: Task 2 approach to developing an enhanced core catcher design.

As shown in Figure 5, the core catcher design proposed in this INERI consists of several interlocking sections that are machined to fit together when inserted into the lower head. The use of interconnected sections of the core catcher reduces manufacturing costs and simplifies installation. Each section of the IVCC consists of two material layers with an option to add a third layer (if deemed necessary): a base material, which has the capability to support and contain the mass of core materials that may relocate during a severe accident; an oxide coating material on top of the base material, which resists interactions with high-temperature core materials; and an optional coating on the bottom side of the base material to prevent any potential oxidation of the base material during the lifetime of the reactor. A combination of scoping coolant flow, thermal, and structural analyses and scoping materials interaction tests were used to identify IVCC materials, thickness, and placement in the RCS. Results suggest that the base material should be either carbon steel or a stainless steel, such as SS 304. However, the use of stainless steel was initially pursued because it would preclude the need for a corrosion-resistant undercoating on lower surface of the IVCC. Evaluation efforts suggest that an insulator coating should be applied on the IVCC upper surface using thermal plasma spray techniques. Although several coatings appear viable, results suggest that the insulator coating should consist of a 500 μm thick ZrO_2 coating over a 100-200 μm thick bond coating of Inconel 718.

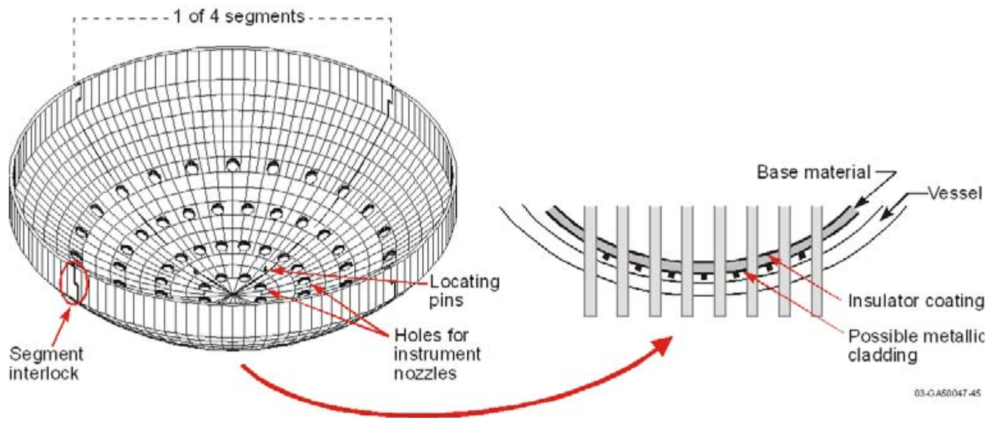


Figure 5: Conceptual design of proposed APR1400 IVCC design.

Evaluations completed in this INERI included materials interactions tests at INEEL's HTTL, simulant tests in LAVA-GAP facility, and tests with prototypic materials at INEEL's HTTL. Materials interactions tests (see Figure 6) were used to verify the high temperature performance of materials proposed for the IVCC coating and base material in steam environments. A series of six tests were completed in the 1/8th scale LAVA-GAP facility (see Figure 7) to investigate the performance of an IVCC when exposed to relocating high temperature simulant materials. Last, two tests were completed in the HTTL using a simulated IVCC exposed to molten prototypic corium materials in inert and oxidizing conditions (see Figure 8).

Results suggest that the proposed IVCC concept is viable and will reduce heat loads to the vessel for a range of severe accident conditions. However, it should be noted that only preliminary IVCC design and evaluations could be completed in this INERI. More detailed analyses and testing are needed before an IVCC could be implemented into a reactor. In particular, tests are needed to confirm the long-term endurance of proposed materials to hydrodynamic loads during operating and accident conditions and assess the impact of the IVCC on coolant flow phenomena such as pressure drop and mixing. In addition, confirmatory tests of irradiation and coolant chemistry effects on coating performance may be warranted.

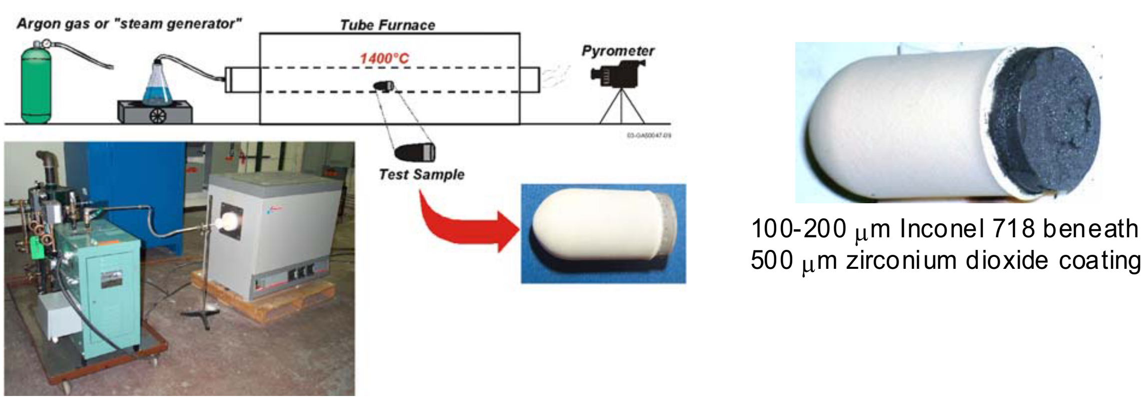


Figure 6: Setup for high temperature materials testing with sample before and after testing.

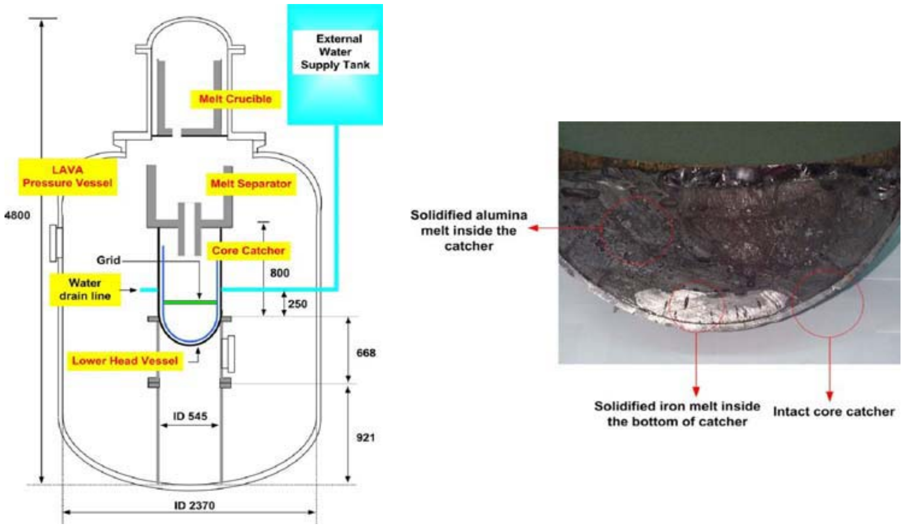


Figure 7: Schematic diagram of the LAVA-GAP facility (dimensions in mm) with LAVA-GAP-3 test section.

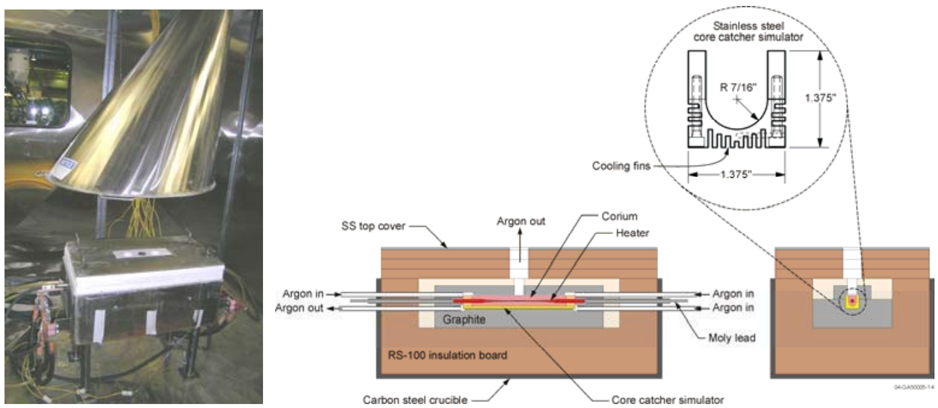


Figure 8: HTTL test assembly used for prototypic testing.

To evaluate the benefit of the IVCC, a correlation was developed to quantify heat transfer in the narrow gap between the IVCC and the reactor vessel lower head developed using results from tests conducted in the GAMMA 1D, GAMMA 2D, and DELTA 3D facilities. Correlations for the entire boiling curve were developed using these one-dimensional, two-dimensional, and three-dimensional test facilities. The GAMMA 1D facility (see Figure 9) also allowed complex flow patterns to be visualized for various heater surface orientations, channel lengths, and gap sizes. The GAMMA 2D tests allowed narrow gap heat transfer to be investigated as a function of gap size and pressure. DELTA 3D facility results provided insights about the vapor film thicknesses in pool boiling. As noted in Rempe, *et al*. (2005), data obtained from these investigations are similar to data recently obtained from other investigations of narrow gap cooling. **(Kune: Reynold's number? Would you rather go back to the table & text from the draft?)**

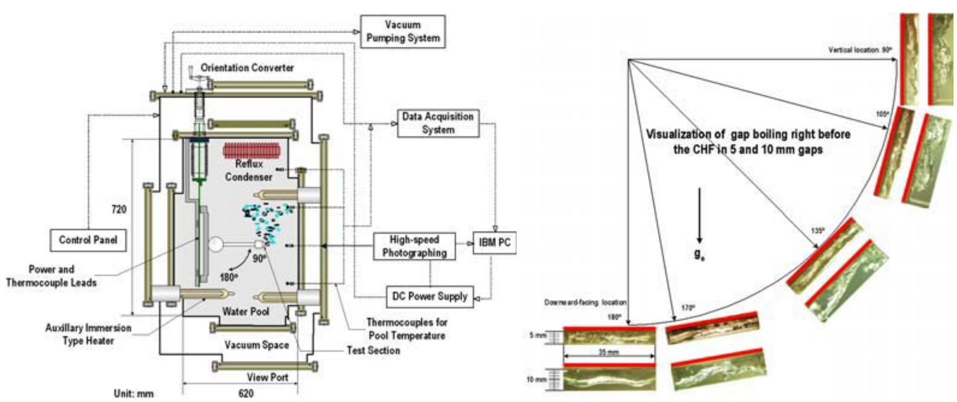
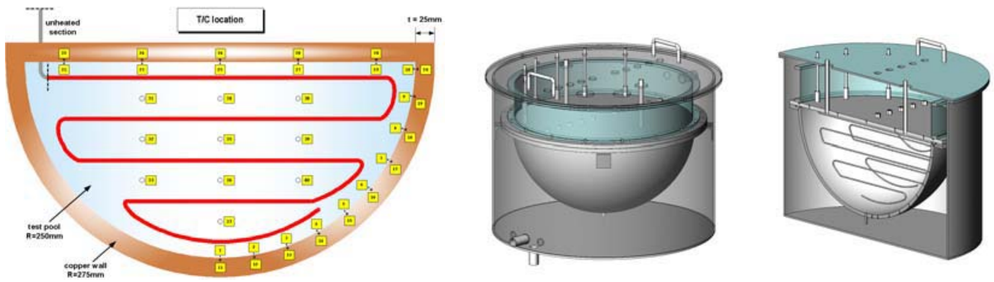


Figure 9: GAMMA 1D provided key insights and data for developing narrow gap boiling heat transfer curve.

Tests were also completed to quantify the heat load from relocated corium to the IVCC. As noted in Rempe, *et al*. (2005), it was not clear that results from previous investigations of turbulent natural convection in an internally heat pool were applicable to the molten pools that may form during a severe accident for higher power reactors. For this project, tests were conducted in SIGMA 2D, which is a two-dimensional semicircular pool with diameter, height, and width of 500 mm, 250 mm, and 100 mm and in SIGMA 3D, which is a three-dimensional, hemispherical pool employing the same internal heating method used in SIGMA 2D (see Figure 10). Data were obtained for quantifying the heat split fraction and angular heat flux. In general, data from these facilities yielded results for the heat split fraction and angular heat flux that were similar to that obtained from previous studies in the literature.



(a) SIGMA 2D (b) SIGMA 3D

Figure 10: SIGMA 2D and SIGMA 3D tests sections.

2.3 Task 3 ERVC enhancement development and evaluation

In Task 3, various types of experiments were performed at SNU, PSU, and KAERI to obtain essential information about ERVC phenomena. Key issues that were considered in Task 3 included delayed flooding of the reactor vessel owing to the large volume between the cavity floor and the lower head, local CHF limits for downward facing boiling on the vessel outer surface, thermal margin for maintaining the integrity of the reactor vessel, methods for ERVC enhancement, two-phase natural circulation through the gap between the reactor vessel and the insulation structure, and choking limit for steam venting during ERVC. To provide insights for addressing these key issues, boiling and two-phase flow tests were conducted in the DELTA and GAMMA facilities at SNU, the SBLB facility at PSU, and the HERMES-HALF facility at KAERI.

Facilities, such as the SBLB, allow steady state boiling experiments to be conducted on simulated reactor vessels. As noted in Rempe *et al*. (2005), the SBLB test facility was advantageous compared with other existing facilities for obtaining CHF in this program because the test section can easily be exchanged (e.g., plain vessels, vessels with coatings, vessels with enhanced vessel/insulations designs can easily be tested). Figure 11 shows a pictorial view of the SBLB water tank with an enhanced vessel/insulation design. As shown in Figure 12, this design involves the use of a non-uniform gap size for the bottleneck by opening the space available for steam venting in the circumferential locations away from the four shear key positions. In so doing, the cross-section flow area of the annular channel at the bottle neck is substantially increased, thus reducing the superficial velocity of steam through the bottleneck. This allows a much higher mass flow rate of steam that could be generated from higher heat fluxes associated with relocated corium. SBLB tests indicate that proposed enhancements could significantly increase the heat that can be removed from a vessel submerged in water (bcause choking limits for steam venting (CLSV) and CHF limits are increased). For example, SBLB results shown in Figure 13 illustrate that a local CHF enhancement of 200% to 330% over a plain vessel could be achieved using an enhanced insulation structure with microporous vessel coatings.

Figure 11: Pictorial view of SBLB water tank with APR1400 vessel/insulation simulator.

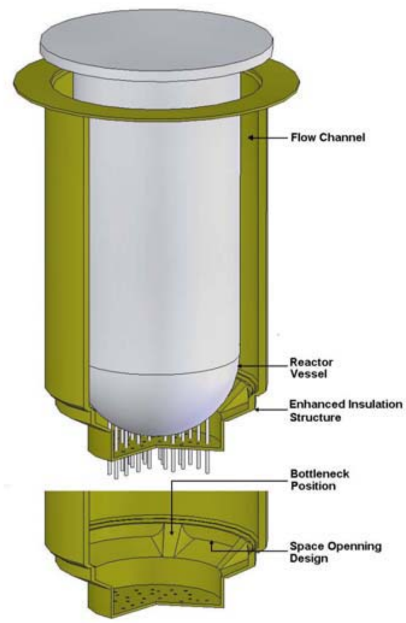


Figure 12: Schematic of enhanced APR1400 vessel/insulation design. (Bill, please send me a jpg or pic of this.)

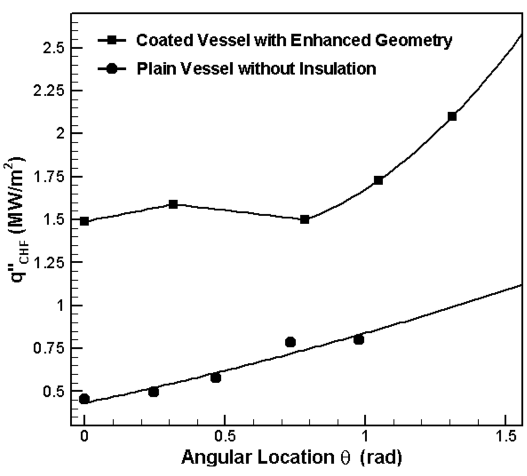


Figure 13: Variations of the local CHF limit on the vessel outer surface for cases with and without vessel coating and enhanced insulation.

HERMES-HALF non-heating (air/water) experimental tests were completed to observe and evaluate two-phase natural circulation through the annular gap between the reactor vessel and the insulation. Flow observations indicate that choking flows occurred in the region near the shear keys (see Figure 14). In cases with larger air injection rates and smaller outlet areas, higher recirculation flows were observed in the region near the shear keys. Because of choking phenomena, periodic air back flow occurred near the minimum gap region under conditions with higher air injection rates. Therefore, tests indicated that design modifications in the minimum gap region are required to ensure sufficient flow through the gap. However, to maximize the heat removal capability by enhancing the re-circulation flow rate through the reactor vessel/insulation gap, more extensive experimental results are required to evaluate a wider range of design parameters, such as the inlet position and outlet location. Also, a detailed three-dimensional flow analysis is necessary to evaluate the local effect, such as choking phenomena near the shear key. Furthermore, non-heating experimental results should be compared with heating experimental results to verify the applicability of HERMES-HALF test results.

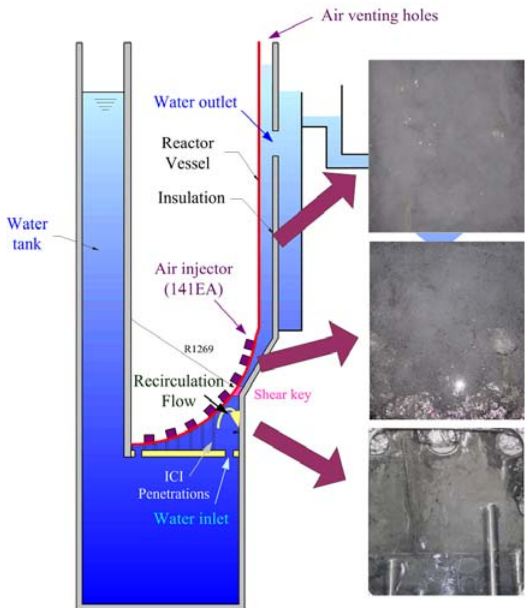


Figure 14: Visualization results of the HERMES-HALF experiment with a 10% (838 m^3/hr) air injection rate of the maximum air flow rate suggested by the IVR evaluations.

2.4 Task 4 calculations to evaluate increased margin with IVR enhancements

In Task 4, the impact of Task 2 and 3 IVR design recommendations was assessed by re-evaluating Task 1 scenarios considering proposed methods to enhance core catcher and ERVC performance. In these evaluations, INEEL applied the SCDAP/RELAP5-3D© and VESTA codes (Rempe *et al*., 1997). KAERI applied the LILAC (Reference?) and RELAP5 codes (INEEL, 1995) and a lumped parameter model (Reference?).

In VESTA calculations, statistical distributions for the heat flux to the vessel wall from several assumed debris configurations were compared with statistical distributions calculated for the CHF from the submerged vessel surface. VESTA uncertainty distributions are Bayesian distributions, which are ultimately combined by a Monte Carlo sampling to yield a distribution on the probability of vessel heat fluxes exceeding the CHF (or if the vessel isn't submerged, the heat transfer rate from the vessel to the reactor cavity). VESTA can consider several types of debris configurations, decay heat power production associated with actinide and fission product heating, heat sources in the metallic material, and uncertainty distributions for a wide range of parameters. A key benefit of the VESTA code is that it allows users to consider the impact of input parameter uncertainties. Hence, VESTA can be applied to obtain point estimates and probability density functions (pdfs) for desired outputs. Heat transfer processes and phenomena modeled by VESTA are described in Rempe *et al.* (1997). The VESTA code was applied to the two debris endstates shown in Figure 15. Configuration A is a stratified configuration with an upper metallic layer; and Configuration C is a stratified configuration with a lower metallic layer. Several types of cases were evaluated for each debris endstate.

- A base case without an IVCC or enhanced ERVC.
- Bases case sensitivities to assumed steel relocation mass.
- Sensitivities that simulate the use of an IVCC.

- Sensitivities to various types of enhanced ERVC.

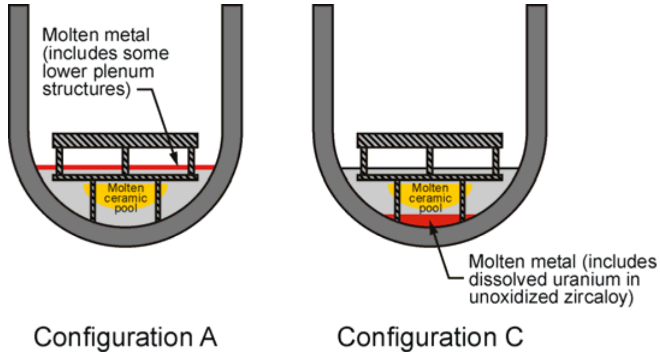
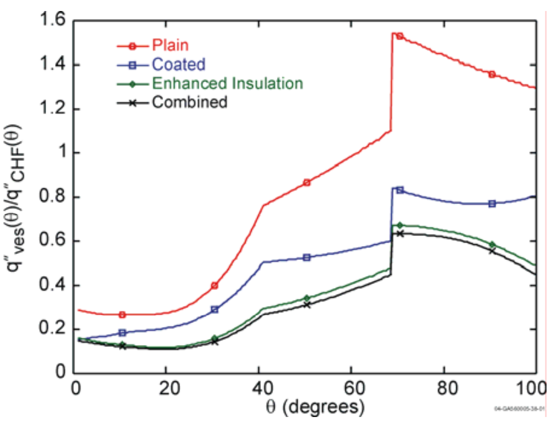


Figure 15: Debris configurations evaluated for the APR1400.

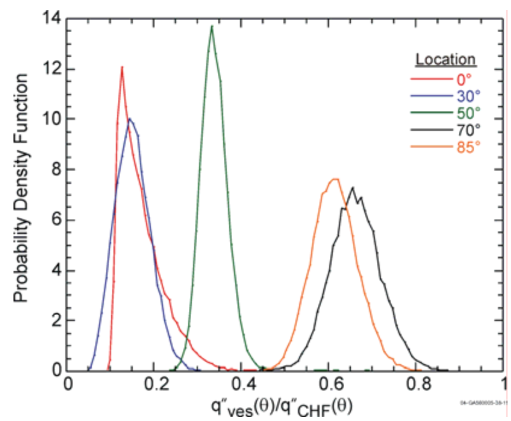
VESTA results for base case conditions in a Configuration A endstate indicate:

- Peak heat fluxes and associated CHF ratios occur near the top of the ceramic pool beneath the metallic layer.
- A small, but non-zero, probability exists for the CHF ratio to exceed 1.0.
- An IVCC must decrease heat fluxes by at least a factor of 2 for this configuration. For example, if the IVCC prevents melt relocation for at least 4 hours, it will maintain vessel heat fluxes below the CHF (and prevent vessel failure).
- ERVC enhancements are estimated to decrease heat fluxes by factors of 2 to 5. Either of the explored ERVC enhancements are sufficient to reduce vessel heat fluxes below the CHF. If one considers uncertainties, the reduction offered by an enhanced insulation design in combination with a coated vessel is required to maintain heat fluxes below CHF.

Point estimate results in Figure 16(a) illustrate the increased margin offered by proposed ERVC enhancements. As shown in Figure 16(b), the pdfs predicted for Configuration A for a case with an enhanced vessel/insulation configuration indicate that the vessel heat flux remains below the CHF.



(a) Point estimate values for ERVC enhancements



(b) Uncertainty analysis for enhanced insulation/vessel configuration

Figure 16: Configuration A results for cases with proposed ERVC enhancements.

VESTA results for base case conditions in a Configuration C endstate suggest:

- Higher heat fluxes occur at vessel locations adjacent to the ceramic layer.
- CHF ratios peak at two locations for this configuration: near the bottom center of the vessel and near the top of the ceramic pool.
- Either an IVCC, which can prevent relocation onto the vessel for at least 4 hours, or a vessel with the combined ERVC enhancements considered in this study are sufficient to maintain vessel heat fluxes below the CHF.

In summary, VESTA results suggest that any of the proposed IVR enhancements maintain vessel heat fluxes below the CHF. Depending upon the selected IVR enhancement and debris endstate configuration, IVR margins may increase by factors ranging from two to four.

Results from calculations performed with all of these methods (SCDAP/RELAP5-3D©, VESTA, LILAC, RELAP5, and the lumped parameter model) indicate that proposed enhancements are needed to provide additional margin for IVR when the vessel is subjected to Task 1 bounding heat loads. In many of the cases evaluated, the enhanced cooling associated with a coated reactor vessel or an enhanced vessel/insulation configuration was sufficient to reduce heat fluxes below CHF. Even greater margins for IVR were predicted for cases with both a coated vessel and an enhanced vessel/insulation configuration. Analyses also suggest that significant additional cooling is possible with an IVCC.

It should be recognized that these calculations were an initial application of these methods to the extreme conditions proposed in Task 1 scenarios. In particular, these initial applications indicated that several model parameters applied in previous calculations were not applicable to these severe conditions. Additional evaluations are recommended that consider other debris endstates, a broader range of sensitivity studies, more detailed IVCC evaluations, and model improvements for molten pool heat transfer and vessel melting. Calculations identified several areas where modeling tools should be revised and additional data are needed for simulating these conditions. For example, in the case of the SCDAP/RELAP5-3D© code, additional data and evaluations are needed for simulating the heat transfer between the molten pool, its crust, and the vessel for conditions where the heat load is sufficient to cause significant crust thinning.

3. CONCLUSIONS AND RECOMMENDATIONS FOR FURTHER EVALUATION

As part a three year INERI, state-of-the-art analytical tools and key U.S. and Korean experimental facilities were used to explore options, such as enhanced ERVC performance and the use of internal core catchers, that have the potential to ensure that IVR is feasible for high power reactors. This increased margin has the potential to improve plant economics (owing to reduced regulatory requirements) and increase public acceptance (owing to reduced plant risk). Although this program focused upon the Korean Advanced Power Reactor - 1400 MWe (APR1400) design, recommendations were developed so that they can easily be applied to a wide range of existing and advanced reactor designs.

Key results, conclusions, and recommendations for future evaluations for activities completed in this program are highlighted below.

- **IVR Bounding Scenario Selection** – To obtain quantify representative late-phase melt conditions that could affect the potential for IVR of core melt following a severe accident in the APR1400, the SCDAP/RELAP5-3D© and SCDAP/RELAP5/MOD3.3 codes were applied to the APR1400 plant. Although an extensive series of severe accident calculations is required to identify bounding transients, Loss of Coolant Accidents (LOCAs), Station BlackOuts (SBOs), and Loss of FeedWater (LOFW) scenarios were assumed to be major IVR scenarios. Accordingly, a cold leg break (representing the LOCA response) and an SBO with LOFW (to combine remaining dominant IVR scenarios) were selected for analysis. Predicted values for vessel failure time, hydrogen generation, melt relocation masses, melt relocation volumes, decay heat in the relocated corium, and power densities in the relocated corium were compared. For the cases of interest, values predicted by SCDAP/RELAP5/MOD3.3 and SCDAP/RELAP5-3D© were similar. Regardless of the transient considered, results for all calculations led to predictions of large melt masses (~100,000 kg total, or more) relocating at high temperatures (~3,000 K, or higher). These results appear to be consistent with the nature of the transients considered. Specifically, all cases involved complete core dryout and subsequent core heatup in a steam environment. Protracted periods (~1 h, or more) of complete core uncovery were sustained in each calculation, leading to development of large core melt masses at temperatures well above the fuel liquidus.
- **IVCC Design and Evaluation -** Although new experimental and analytical investigation results suggest that an in-vessel core catcher is viable and can significantly reduce the heat loads to the vessel from relocated core materials, more detailed evaluations are needed to fully assess its merit.

As discussed above, scoping calculations indicate that the presence of the in-vessel core catcher will not significantly impact coolant flow. However, confirmatory thermal-hydraulic tests should be completed to assess the impact of the IVCC on coolant flow prior to its inclusion in the RCS. In addition, more detailed experimental evaluations are needed to confirm that proposed coatings will not degrade over the lifetime of the reactor. It is also suggested that experimental evaluations be conducted to assess in-vessel core catcher performance when subjected to sustained heat loads. Such evaluations could be conducted by placing heat sources in the simulant materials that relocate in the LAVA-GAP facility. Finally, as noted above, narrow gap heat transfer models should be applied to predict heat transfer observed in recently-completed LAVA-GAP tests.

- **IVCC Narrow Gap Heat Transfer –** The ability to predict heat transfer from a narrow gap has several applications in assessing the potential for IVR. It is needed to predict heat transfer in cracks that develop within relocated core materials, between relocated core materials and the vessel or "engineered gap" that may occur if an in-vessel core catcher is placed within the reactor vessel. Significant progress has been made toward developing a boiling curve for predicting heat transfer in narrow gaps with CCFL. Results indicate that surface angle, gap size, pressure, and dimensional effects must be considered. Data from tests completed in several facilities were used to develop a narrow gap heat transfer model that consider these effects. However, additional evaluations of these models are needed. In particular, they should be applied to predict heat transfer in relocated corium materials and in the gap between an in-vessel core catcher and the vessel. If simulant test data are used, additional material properties must be obtained so that differences between prototypic and simulant material behavior may be better understood.
- **ERVC Enhancements** – Several investigations were completed that explore methods to enhance the margin associated with ERVC. Results show that the efforts to enhance the insulation configuration placed around the vessel structure can significantly improve the steam venting process (and subsequently increase local CHF values). In addition, tests show that coatings can be applied to the vessel external surface (and enhance its coolability). Finally, tests were completed to evaluate the combined effect of enhanced insulation and vessel coatings, and results indicate that the combination of these methods further enhances vessel coolability (but that the combined effect is less than the sum of the individual effects). Correlations were develop and applied for predicting CHF with an enhanced vessel/ insulation configuration, vessel coatings, and a coated vessel with an enhanced vessel/insulation configuration. However, it should be noted that these evaluations represent an initial study for possible enhancements. To maximize the benefits of enhanced ERVC, additional studies are suggested to further optimize parameters associated with the vessel/insulation design and vessel coatings.
- **IVR Margin Assessment** – Results from calculations performed with several methods (SCDAP/RELAP5-3D©, VESTA, LILAC, RELAP5, and the lumped parameter model) indicate that proposed enhancements are needed to provide additional margin for IVR when the vessel is subjected to bounding heat loads. In many of the cases evaluated, the enhanced cooling associated with a coated reactor vessel or an enhanced vessel/insulation configuration was sufficient to reduce heat fluxes below CHF. Even greater margins for IVR were predicted for cases with both a coated vessel and an enhanced vessel/insulation configuration. Analyses also suggest that significant additional cooling is possible with an IVCC. However, it should be noted that these calculations were an initial application of these methods to the extreme conditions proposed in Task 1 scenarios. In particular, these initial applications indicated that several model parameters applied in previous calculations were not applicable to these severe conditions. Additional evaluations are recommended that consider other debris endstates, a broader range of sensitivity studies, more detailed IVCC evaluations, and model improvements for molten pool heat transfer and vessel melting. Calculations identified several areas where modeling tools should be revised and additional data are needed for simulating these conditions. For example, in the case of the SCDAP/RELAP5-3D© code, additional data and evaluations are needed for simulating the heat transfer between the molten pool, its crust, and the vessel for conditions where the heat load is sufficient to cause significant crust thinning.

Clearly, significant progress has been made to improve our understanding of mechanisms that impact IVR. Evaluations completed in this study indicate that the proposed IVR enhancements significantly increase IVR margins for higher power reactors. Additional evaluations suggested as follow-on tasks are designed to optimize these enhancements, ensure their long-term endurance, and improve methods for simulating their performance.

NOMENCLATURE

ALWRs	Advanced Light Water Reactors
APR1400	Advanced Power Reactor - 1400 MWe
CLSV	Choking Limit for Steam Venting
CCFL	Counter-Current Flow Limitation
CHF	Critical Heat Flux
CHFG	Critical Heat Flux Gap
DELTA	Downward Ebullient Laminar Transition Apparatus
ERVC	External Reactor Vessel Cooling
GAMMA	Gap Apparatus Mitigating Melt Attack
HERMES-HALF	**Definition?**
HTTL	High Temperature Test Laboratory
INERI	International Nuclear Energy Research Initiative
INL	Idaho National Laboratory
IVCC	In-Vessel Core Catcher
IVR	In-Vessel Retention
KAERI	Korea Atomic Energy Research Institute
LAVA-GAP	**Definition?**
LILAC	**Definition?**
LOCA	Loss of Coolant Accident
LOFW	Loss of FeedWater
PSU	Pennsylvania State University
RELAP5	Reactor LOCA Analysis Program Version 5
SBLB	Subscale Boundary Layer Boiling
SBO	Station BlackOut
SCDAP	Severe Core Damage Analysis Program
SIGMA	Simulant Internal Gravitated Material Apparatus
SNU	Seoul National University
TMI-2	Three Mile Island Unit 2
US NRC	U.S. Nuclear Regulatory Commission
VESTA	VEssel Statistical Thermal Analysis

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